NON-PUBLIC?: N
ACCESSION #: 9201090147
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000220

TITLE: Reactor Scram Due To Equipment Failure
EVENT DATE: 12/04/91 LER #: 91-014-00 REPORT DATE: 12/26/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 097

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: W. Drews TELEPHONE: (315) 349-2444
Technical Support Manager NMP1

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JB COMPONENT: FI MANUFACTURER: G080
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 4, 1991, at 0842 hours, with the mode switch in "RUN" and reactor power level at approximately 96.5% of rated, the Nine Mile Point Unit 1 (NMP1) Reactor experienced a reactor scram. The reactor scram was the result of a level transient that lowered reactor water level below the scram setpoint of greater than or equal to 53 inches.

The root causes were manufacturing defects and design inadequacies. The immediate cause of the event was equipment failure. A solder connection failed on the total steam flow meter in the Feedwater Level Control circuitry, causing an imbalance in the signals to the 3 element controller, which closed 13a and 13b Feedwater Flow Control Valves in response.

Initial corrective actions were to respond to the reactor scram and increase reactor water level to its normal range. The cause of the level

transient was investigated, and the failed total steam flow meter was identified and replaced. The remaining meters in th
Feedwater Control
System were evaluated and those without electrical shunts were replaced. An evaluation will be performed of meters in other control circuits to prevent similar events.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On December 4, 1991, at 0842 hours, with the mode switch in "RUN" and reactor power level at approximately 96.5% of rated, the Nine Mile Point Unit 1 (NMP1) reactor experienced a reactor scram. The reactor scram was the result of a level transient that lowered reactor water level below the scram setpoint of greater than or equal to 53 inches on both channels of the Reactor Protective System (RPS).

Prior to reaching the RPS water level scram setpoint (greater than or equal to 53 inches), control room operators attempted to take manual control of the Feedwater Control System. Upon receipt of the scram signal, all control rods inserted to position 00. The High Pressure Coolant Injection System (HPCI) initiated on low reactor water level (greater than or equal to 53 inches), and increased level to approximately 100 inches,as expected. All equipment operated as designed, except that which is described below. The lowest reactor water level during the event was approximately 17 inches indicated (approximately 8.5 feet above active fuel).

The post scram review identified that Control Rod 18-07 had a slow 5% scram insertion time of .405 seconds, instead of the Technical Specification required less than .398 seconds. The 20%, 50%, and 90% scram insertion times for Control Rod 18-07 are slower than average, but are within acceptable ranges. This control rod has shown signs of seal degradation (high stall flow rates and high temperatures). Work Request (WR) 188791 was issued to troubleshoot this Control Rod.

Also during this event, a Turbine Building Fire Alarm annunciated in the Control Room, which caused operators to respond. This alarm, which is associated with the Turbine Generator Trip relay (86G2), indicates that the Fire Protection Water Deluge System for the Main Transformer is in the "permissive to operate" mode. This alarm has been previously identified as a nuisance alarm, and a Modification Request (N1-88-137) had been initiated to correct.

The review of the Post-Trip Log and Balance of Plant Log revealed that several computer points had insufficient scan rates to facilitate useful trending data for pre and post trip parameters. This data is not considered critical to the evaluation of the event, but WR 188298 was written for the Computer Department to change the scan rate of 6 computer points for future enhancements.

II. CAUSE OF EVENT

A root cause analysis was performed using Nuclear Division Procedure NDP-16.01, "Root Cause Evaluations."

The root causes of the event were attributed to manufacturing defects and design inadequacies. Metallurgical analysis determined that the solder joint of the total steam flow meter was not heated to an adequate temperature to assure the flux was sufficiently expelled from the joint. A significant contributor to this event was design inadequacy. The failed solder joint lacked a mechanical assembly to hold the wire prior to soldering. The failed meter was original equipment installed in the plant.

Using the Data Acquisition and Analysis System (DAAS), (which had been installed to monitor perturbations in the Feedwater System), the System Engineer isolated the apparent cause of the transient to a malfunction of the 3 element feedwater level controller. Investigation revealed that a "zero" steam flow signal occurred approximately 20 seconds prior to the reactor scram. This signal was caused by an electrical lead on the Steam Flow meter (located on E console in the Control Room), becoming disconnected from its solder point. When this occurred, a large flow error signal was generated, resulting in the 3 element controller sending a closure signal to 13a and 13b Feedwater Flow Control Valves. When the two Feedwater Control Valves closed, Reactor Water Level decreased to below the scram setpoint of greater than or equal to 53 inches.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR 50.73(a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineering Safety Feature (ESF), including the Reactor Protection System (RPS)."

During normal operation, the Level Control System will add individual Feedwater Flow signals (FF) and Steam Flow signals (SF), to arrive at total FF and SF signals. The total FF and SF are compared to each other to provide a Flow/Error. The Flow/Error signal is then compared to the Level signal to provide a Level/Flow signal. During steady state operation, Feed Flow and Steam Flow are equal and the resultant Flow/Error is zero (0), therefore, the Level/Flow signal consists only of the Level input. The level input is compared to the desired level set by the control room operator, and a signal is sent to the Feedwater Control Valves for correct valve position to maintain reactor level. If any of the three parameters change during operation, an error signal is generated, and the system will react to change the valve position to restore steady state conditions.

III. ANALYSIS OF EVENT (cont.)

On December 4, 1991, an electrical lead in the Total Steam Flow Meter became disconnected from its solder point. This caused the Level Control System to see zero Steam Flow. When this occurred, a large Flow/Error signal was generated. The system reacted by sending a signal to close the Feedwater Control Valves to respond to the sensed loss of Steam Flow. Feedwater Flow decreased, causing a corresponding decrease in reactor level. The system functioned as designed for the sensed parameters.

There were no adverse safety consequences as a result of this event as all systems and equipment required to safely shut down the reactor operated as expected. This event is bounded by the transient analysis in the Final Safety Analysis Report (FSAR) Chapter XV, Section 3.13, "Feedwater Controller Malfunction (Zero Demand). The FSAR transient assumes that there is no make-up water from any source, and a Main Steam Isolation Valve (MSIV) isolation in approximately 25 seconds.

IV. CORRECTIVE ACTION

Immediate corrective action involved responding to the scram per the applicable procedures, to safely shutdown and cooldown the reactor. Also, an investigation into the cause of the scram was initiated per procedure N1-REP-6, "Post Reactor Scram and Evaluation."

Additional corrective actions include:

1. Replacing the Total Steam Flow Meter with a new meter, which has a shunt provided across the input and output terminals to prevent reoccurrence of a "Zero" output signal, if the electrical lead is disconnected.

2. A review of the remaining meters in the Feedwater Control System was conducted to ensure that shunts were installed or solder connections were intact. Meters without shunts were evaluated and replaced as necessary. Meters for reactor water level, reactor pressure, and total feedwater flow were replaced.

3. A review of meters installed in control circuits of the following systems will be completed to ensure shunts are installed or solder connections are intact. These meters will be replaced as necessary.

o Reactor Recirculation Control o Core Spray
o Reactor Level, Reactor Pressure o Containment Spray
o Reactor Building Emergency Ventilation o Reactor Water Clean-up
o Emergency Condensers o Control Rod Drive

V. ADDITIONAL INFORMATION

A. Previous similar events:

There are three previous Licensee Event Reports (LER) that are due to Low Reactor Water Level. These events are discussed in LER's 85-17, 85-22, and 87-31. Although these events are similar, they are not caused by a failure of the Feedwater Level Controller. Therefore, the corrective actions in these reports would not have prevented the occurrence of this event.

B. Failed components:

COMPONENT IEEE 803 FUNCTION IEEE 805 SYSTEM ID

Total Steam Flow Meter FI JB
Feedwater Control System LC JB

C. Identification of components referred to in this LER:

COMPONENT IEEE 803 FUNCTION IEEE 805 SYSTEM ID

Reactor Level (E-Panel) LI JB
Total Feedwater Flow FF JB
Reactor Pressure PI JB
Control Rod 18-07 NA AA
Computer System NA ID
Fire Detection FRA IC

Other corrective actions arose from problems identified during this event, and are provided for information only.

1. WRs 188791 and 188298 were written to troubleshoot problems identified with Control Rod 18-07 and to change scan times for computer points for data acquisition.

2. Modification Request N1-88-137 is being re-evaluated for work to prevent nuisance alarms on the Fire Protection Deluge System.

ATTACHMENT 1 TO 9201090147 PAGE 1 OF 1

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Joseph F. Firlit
Vice President
Nuclear Generation NMP83177

December 26, 1991

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-220
LER 91-14

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 91-14 Which is being submitted in accordance with 10CFR50.73 (a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System".

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Joseph F. Firlit
Vice President - Nuclear Generation

JFF/MD/Imc
ATTACHMENT

xc: Thomas T. Martin, Regional Administrator Region I
Wayne L. Schmidt, Sr. Resident Inspector

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